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NO. 174

"Evaluation of Neutron Dose Measurement Techniques for Use in a Shipboard Environment"

UNITED STATES NAVAL ACADEMY
ANNAPOLIS, MARYLAND

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Recently, the biological effect of fast neutrons has been under review. As a result, adjustments may be required in the current dose equivalence algorithms when fast neutron fields are present. Since many Navy personnel are exposed routinely to fast neutron radiation fields, there is a need to be able to reduce any uncertainties about the amount of exposure that they have received. While many devices are available for personnel neutron dosimetry, an enhanced

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"Evaluation of Neutron Dose Measurement Techniques for Use in a Shipboard Environment"

A Trident Scholar Project Report

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ABSTRACT

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This project explores the relative response of bubble dosimeters to several different types of neutron measurement systems when exposed to three different fast neutron sources. Computer simulations of the neutron source and detectors are also provided to help evaluate the accuracy of the detectors. These data are used to develop comparisons of neutron detector responses for fast neutrons. Correction factors are determined for several of the detectors for 14 MeV neutrons to extend the useful energy range of these devices.

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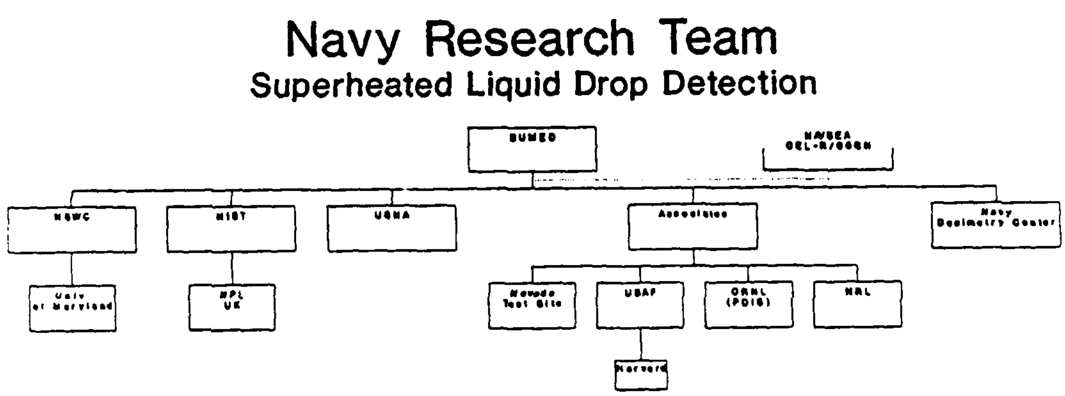
1. INTRODUCTION

Radiation, when interacting with matter, results in direct and indirect mechanisms to produce ionizations. All methods of detecting neutron radiation involve the use of indirect means to quantify the field strength.
The reason for this is that the neutron has no electrical charge and does not directly produce ionizations when it interacts with matter. Therefore neutron dosimetry relies on the products of a neutron induced reaction to produce ionizations from which the absorbed dose or dose equivalent can be determined. Unfortunately, the indirect methods of measurement are sensitive to such factors as the kinetic energy of the incident neutrons, the exposure area's geometry, and the presence of other types of ionizing radiation. Often different methods will produce different answers for the same neutron source. This has led to the development of correction factors to normalize the response of the various devices to the accepted standards.

Many methods of neutron detection have been developed. By comparing the response of different neutron detection methods to various neutron sources, a more accurate calculation of the dose equivalent received by personnel can be made. This project compares the response of the AN/PDR-70 neutron remmeter, the tissue equivalent proportional counter (TEPC), the thermoluminescent dosimeter (TLD), the bubble dosimeter, the CR-39 track etch dosimeter, and the NE-213 liquid scintillator with each other in order to compare their responses to fast neutrons. Ideally, one would like each detector to give the same dose measurement when exposed to the same conditions. To accomplish this, correction factors are determined for each detector such that all of the detectors measure the same dose. Unfortunately these correction factors are spectrum sensitive and will change with the energy of the incident neutrons. An unmoderated Californium (Cf-252) source, a Plutonium-Beryllium (Pu-Be) source, and the USNA 14 MeV neutron generator provided three separate energy points for the determination of fast neutron correction factors.

Figure 1.1 - Navy Research Team

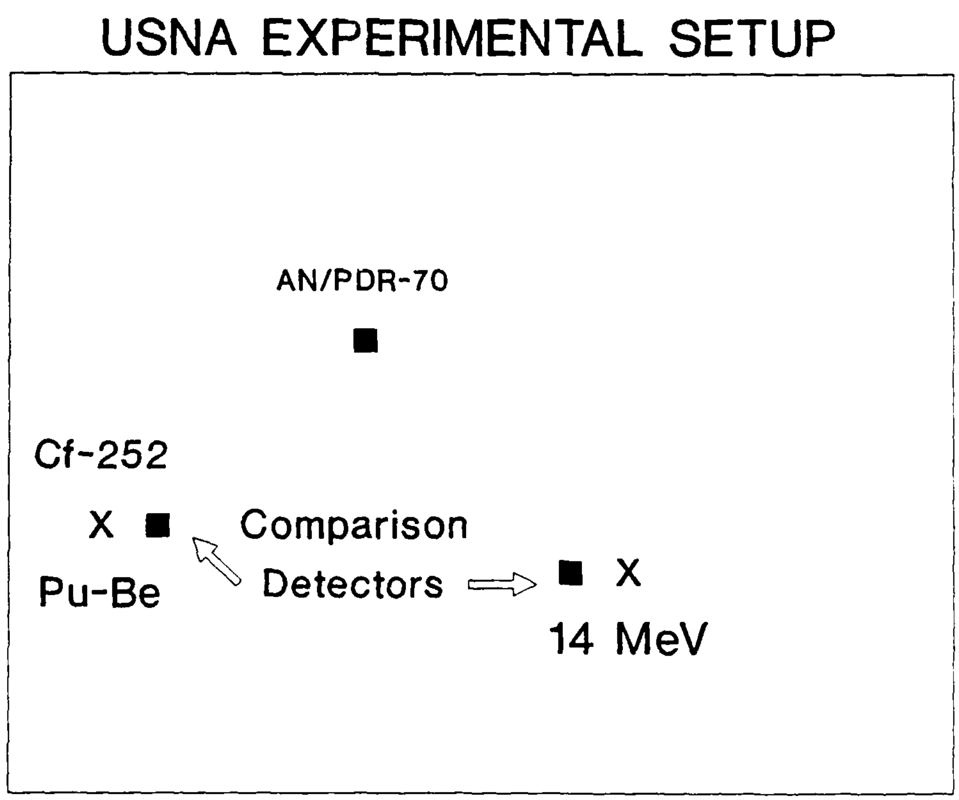


BUMED - Bureau of Medicine and Surgery
NSWC - Naval Surface Warfare Center
NIST - National Institute of Standards & Technology
NAVSEA - Naval Sea Systems Command
NPL - National Physical Laboratory (UK)
USNA - United States Naval Academy
ORNL - Oak Ridge National Laboratory
NRL - Naval Research Laboratory

This research was an integral subset of the work in progress through the Navy's superheated liquid drop detection research team which is diagrammed in Figure 1.1. As can be seen, many different organizations are involved in this project. USNA's primary responsibility has been to study the bubble dosimeter's response compared to the response of other devices. Of particular importance was the response of each device to 14 MeV neutrons, as their response has not been well characterized at this energy. A problem in measuring these fast neutron sources is that neutrons begin interacting with the surroundings and losing their energy as soon as they leave the source. A person desiring to measure fast neutron response must be certain that a significant fraction of the neutrons at a given distance from the source are still fast neutrons. For these experiments, computer modeling was used to help evaluate the fast neutron contribution to the dose at varying distances from the sources used. A distance of 25 cm was used for the 14 MeV neutron generator measurements whereas 20 cm was used for the Cf-252 and Pu-Be sources as a result of the computer predictions. Each detector was then evaluated in separate tests to eliminate any disturbances to the field that one detector could potentially have on another near it. An AN/PDR-70 was used as a normalizing detector to compare all of the device tests to a common instrument. A diagram of the floor plan setup of the test cell is shown in Figure 1.2.

This report has been organized into individual sections about the theoretical principles and experimental technique used with each of the comparison detectors. A discussion of how each device compares for dose calculations, relative response, and room return effects is then presented. The very nature of this report leads to some technical terms used in neutron dosimetry that may be unfamiliar to the reader. To overcome this problem, Appendix A contains definitions of several terms that may be unfamiliar or unclear to the reader.

Figure 1.2 - Experimental Setup of Test Cell



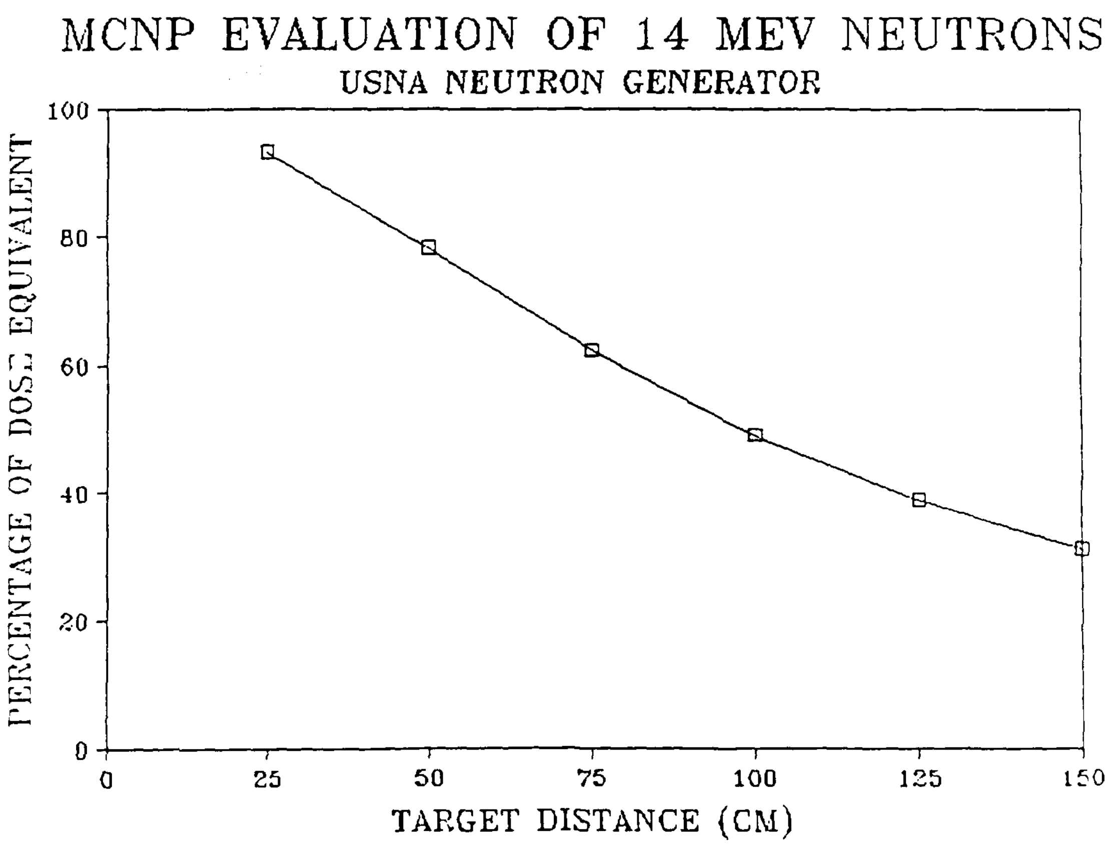
2. COMPUTER CODES

This project used two major computer codes to assist in the planning and understanding of the experimental data. To better predict the neutron energy dependent flux at various points in the test cell the "MCNP - Monte Carlo Neutron and Photon Transport Code System"[2] and the "SAND II - Neutron Flux Spectra Determination by Multiple Foil Activation -- Iterative Method"[3] were employed. Both of these codes allowed for many discrete variables to be analyzed quickly without having to run time-consuming experiments. Once the options had been explored and a narrow area of experimental interest was defined, reliable and useful data were obtained quickly and efficiently.

The MCNP code is a general Monte Carlo code for neutron and photon transport. It first requires the user to define the geometry of the area to be studied by partitioning it into specific closed cells bounded by planes and spheres consisting of user defined materials. Next, neutron sources and detector positions are defined within the cell. Neutron interactions are then generated using random numbers, for as long as the user wishes, with the help of extensive libraries of material cross section data. Each neutron is allowed to undergo collisions, which result in their energy being lost throughout the modeled cells until they leave the simulation area or reach a detector location and are counted. The program then tallies the neutrons that strike the detector and categorizes them into user defined energy bins to get a picture of the fluence spectrum per neutron emitted from the source. The simulation also provides a dose equivalent output by multiplying the fluence in each bin by a dose conversion factor and then summing the results in all energy bins. The code allows for a wide range of sources, detectors, cell setups, and many other factors to provide a very flexible and useful simulation package. It also provides its own calculation of a figure of merit used to approximate the accuracy of a given simulation.

In this project, MCNP was used with both a simulated 14 MeV neutron and Cf-252 source to determine the best location to place the detectors and dosimeters during actual test runs. A simplified model of the USNA nucleonics test cell was defined that included the concrete walls, floor, and ceiling but left out the metallic equipment in the room. The equipment was not included because of the relatively small effect it was felt this equipment would have on neutron interactions and the extreme complications and longer running times that would be caused in specifying all of the equipment present. Since experimental results have agreed reasonably well with the computer simulations, the simplification in cell definition appears to have proven to be appropriate for this study. By varying the distance a theoretical detector was located from the neutron generator, a neutron spectrum in the room was obtained. From this analysis, the percentage of 14 MeV neutron fluence at increasing distances from the generator was calculated. The results, shown in Figure 2.1 indicated that at 25 cm, 94% of the total dose equivalent would be from neutrons whose energies are between 10 and 18 MeV. This was taken as the 14 MeV neutron fluence. A similar analysis was done with the Cf-252 source to determine its optimal detector distance.

Figure 2.1 - MCNP Prediction of 14 MeV Neutron Flux



Dose calculations with the MCNP code were done using the following formula:

$$\dot{H} = S * C_1 * F/C_2 \qquad (2.1)$$

where:

$\dot{H}$ = Dose Equivalent Rate in mrem/hr

S = Source strength in neutrons emitted/sec

C_1 = 3600 sec/hr conversion factor

F = Particle Fluence in Rem/neutron emitted

C_2 = 3.6 (Rem/hr)/(mrem/sec) conversion factor

For the Cf-252 runs, the source strength was found with the basic relationship:

$$S = S_o * \exp(-\lambda_c * t) \qquad (2.2)$$

where:

S_0 = Original source strength in neutrons/sec

λ_c = Decay constant of Cf-252, .262 yrs^{-1}

t = Age of source in years

The USNA Californium source had an original strength of 2.51×10^6 neutrons/sec and was approximately three years old.

For the neutron generator runs, the source strength could not be directly measured, so a series of copper foil activations were performed. When irradiated, copper undergoes the following reaction:

$$Cu^{63} + n \rightarrow Cu^{62} + 2n + .51 \text{ MeV} \qquad (2.3)$$

The .51 MeV gamma photopeak was counted using an ND62 multichannel analyzer (MCA). These data were then analyzed to determine the generator's output using the following equations:[4]

$$N = N^{62} * \lambda^{62} * t_c * \varepsilon * \Omega * f_d * f_c * q * a \quad (2.4)$$

$$N^{62} = \Sigma_{a63} * \phi * V * (1 - \exp(-\lambda^{62} * t) / \lambda^{62} \quad (2.5)$$

$$S = \phi * 4 * \pi * r^2 \quad (2.6)$$

where:

N = Net number of counts under .51 MeV photopeak

N^{62} = Cu^{62} atom density after iradiation time t in atoms/cm^3

λ^{62} = Decay constant of Cu^{62}, .00116 sec^{-1}

t_c = Counting time in seconds

ε = System efficiency, .137

Ω = Solid angle foil presents to the detector

f_d = Correction for Cu^{62} decay between irradiation and counting, $\exp(-\lambda^{62} * t_d)$

t_d = Decay time in seconds

f_c = Correction for Cu^{62} decay while counting, $\dfrac{(1 - \exp(-\lambda^{62} * t_d))}{\lambda^{62} * t_c}$

q = Number of .51 MeV gammas Cu^{62} emits per disintigration, 1.95

a = Isotope Abundance of Cu^{63} in the copper foil, .691

Σ_{a63} = Macroscopic absorption cross section of copper in cm^{-1}, based on = 530mb[4]

ϕ = Neutron flux in foil in neutron/cm^2/sec

V = Volume of Copper foil, cm^3

t = irradiation time in seconds

S = generator yield in neutrons/sec

r = foil distance from generator in cm

The system efficiency was found by using a Na^{22} source of known intensity and comparing the number of counts under its .51 MeV photopeak to the expected number that a source of its intensity should produce. The analysis indicated that the neutron generator output could be expressed as:

$$\frac{\text{14 MeV Generator Output}}{\text{Normalizing 70 Count}} = 1.76 \times 10^6 \frac{\text{neutrons}}{\text{70 Count}} \quad (2.7)$$

The result in equation 2.7, when multiplied by the observed 70 counts and dividing by the count time, produced the 14 MeV source strength for use in evaluating the dose equivalent rate in equation 2.1.

MCNP was developed at Los Alamos National Laboratory and is available through the Radiation Shielding Information Center (RSIC) in Oak Ridge, Tennessee. The U.S. Naval Academy's Gould computer system is currently equipped with version 3A and details about its use are available in Appendix B.

The SAND II code uses an iterative perturbation method for determining neutron flux spectra by multiple foil activations. It requires the user to expose several different activation foils, with or without cadmium covers, to a neutron field and then to calculate the saturated activity per target neutron of the sample. The code allows the user to enter the saturated activity of each foil matched with the specific nuclear reaction involved from the computer library. The user also inputs the approximate type of spectrum expected and then the computer code iteratively adjusts the spectrum and the activities until both agree within the limits of certain user defined criteria. It then displays the differential and integral fluxes and creates logarithmic graphs of the input spectrum and the computed differential and integral spectrums. Accuracy of this method is based on both the number of different foil reactions used and the energy range that each foil measures best.

SAND II was used in this experiment to get a better idea of the neutron flux in the room and as a check on the

Table 2.1 - Foil Activation Data[5]

Foil	With or Without Cover	Reaction	Foil Mass Grams	Observed Gamma KeV	q %	Halflife min.
Zr90	Both	(n,2n)	.3341	511	2.8	4.18
Zr90	Both	(n,2n)	.3341	588	93	4.18
Zr90	Both	(n,2n)	.3341	1508	6.7	4.18
Mg24	Without	(n,p)	.0905	1370	100	901.2
Al27	Both	(n,a)	.7503	844	72	9.45
Al27	Both	(n,a)	.7503	1014	28	9.45
Al27	Both	(n,p)	.7503	1369	100	901.2
In115	Both	(n,g)	.3405	417	30	54.2
In115	Both	(n,g)	.3405	819	17	54.2
In115	Both	(n,g)	.3405	1097	53	54.2
In115	Both	(n,g)	.3405	1293	80	54.2
In115	Both	(n,g)	.3405	2112	16	54.2
Cu63	Without	(n,g)	.4244	511	196	9.78

results of other experiments. By irradiating the different foils as shown in Table 2.1 , some with cadmium covers and some without, a fairly accurate picture of the fast neutron spectrum was obtained. Some of the foils offered more than one gamma photopeak to count or multiple reactions to use in finding the saturated activity. The calculation of saturated activities per target atom was done using the following equation:

$$A_{\infty} = \frac{N*A}{t_c*\varepsilon*\Omega*f_d*f_c*q*a*N_a*m} \qquad (2.8)$$

where:

A_{∞} = Saturated activity per target atom in atom/cm^3/sec/target atom

N = Net number of counts under the photopeak of the element being activated

A = Target atomic weight in grams/mole

N_a = Avogadro's number, 6.02×10^{23} atom/mole

m = Target mass in grams

An input of target atom thickness was also required for foils that included cadmium covers:

$$x = (N_a*\rho*t_{cd}/A) \times 10^{24} \qquad (2.9)$$

where:

x = Target atom thickness in barns^{-1}

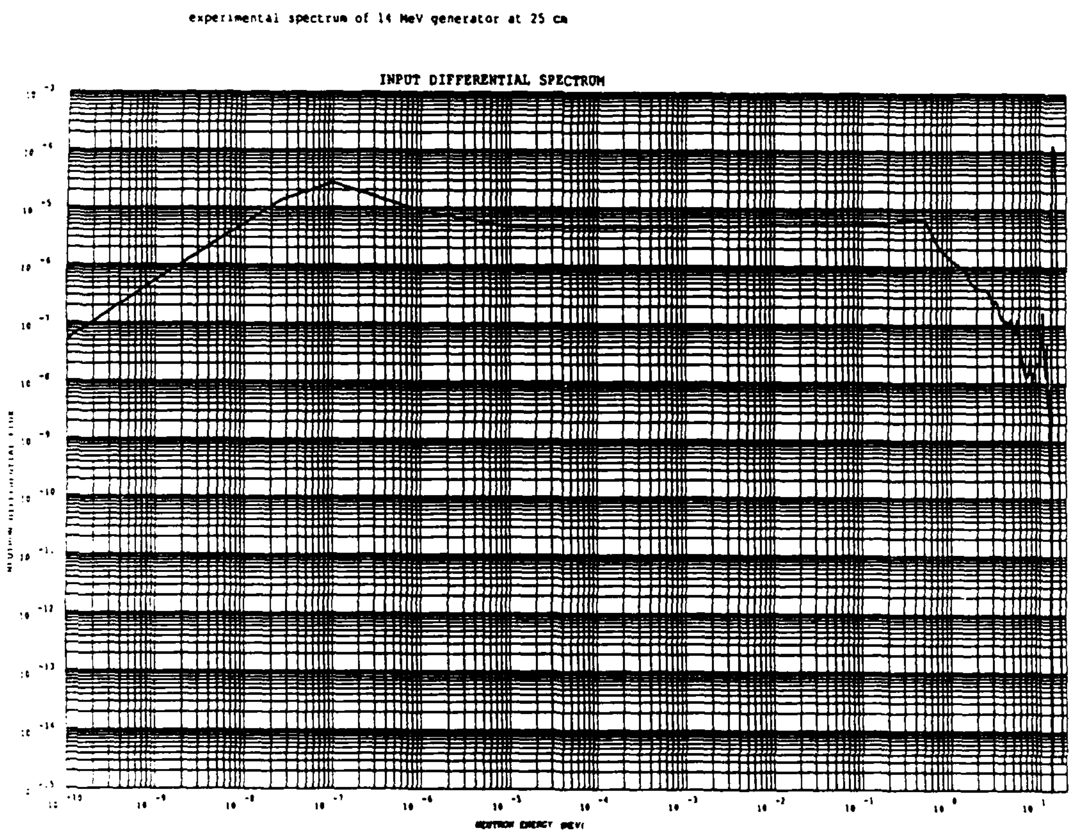
ρ = Target material density in g/cm^3

t_{cd} = Cadmium cover thickness in cm

Figure 2.2 shows the input differential neutron flux at 25 cm which was chosen from the MCNP results. Figure 2.3

shows the output differential spectrum that the SAND II code predicts for experimental runs with the 14 MeV neutron generator. Both of these plots are done on log-log axes and for our fast neutron experiments, the area of interest is primarily the last two decades on the right. The output spectrum successfully predicts the expected 14 MeV peak. The energy range of each foil used is displayed on the plot and it can be seen that our foils primarily respond to fast

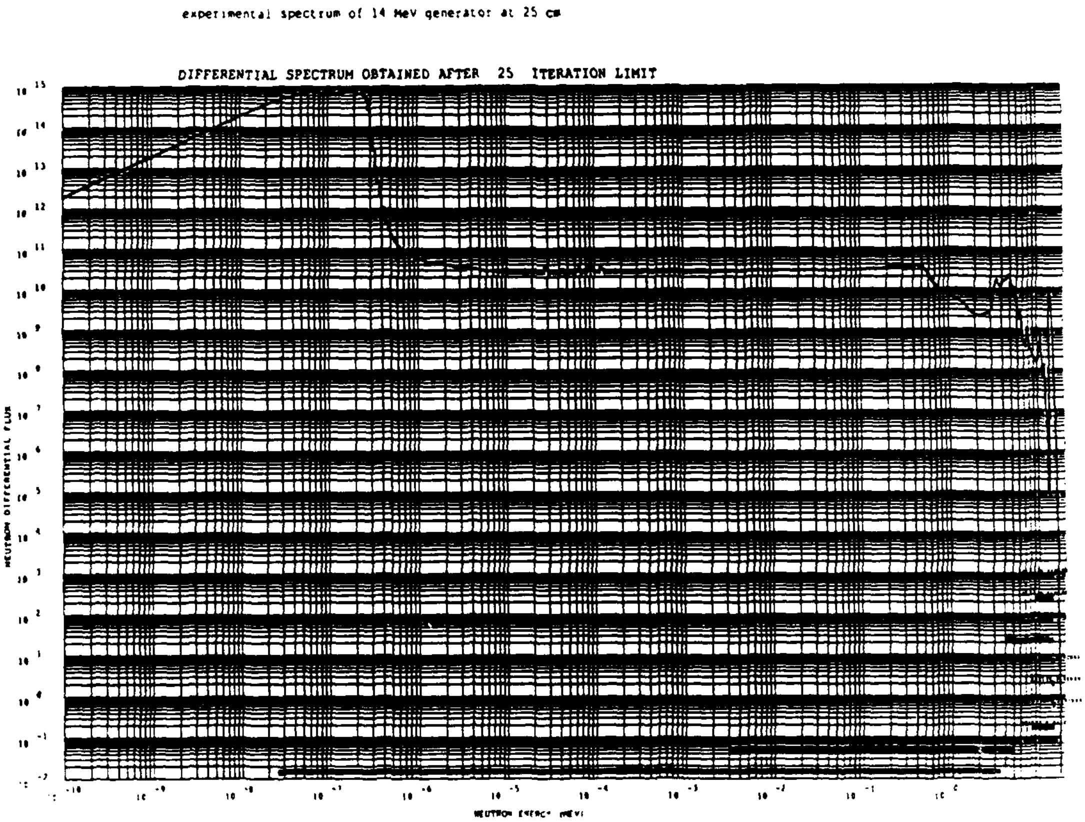
Figure 2.2 - SAND II Input Spectrum of 14 MeV Flux



neutrons. The high response indicated in the thermal range has no experimental validity because none of the activation foils adequately measure this energy region.

SAND II was developed at the Oak Ridge National Laboratory and is also available at the Radiation Shielding Information Center. The U.S. Naval Academy has the 1981 version of the code on its Gould computer system. Details about its specific use are presented in Appendix B.

Figure 2.3 - SAND II Output Spectrum of 14 MeV Flux



3. AN/PDR-70 REMMETER

The AN/PDR-70 radiac set, also known as the snoopy, has been a standard remmeter for neutron measurements aboard naval vessels for many years. The device consists of a boron trifluoride (BF_3) proportional counter surrounded by a boron loaded attenuator and two layers of polyethylene moderator. The counter measures the number of ionizations of the BF_3 gas, which are assumed to be caused by neutron interactions. The counter sends input to a remmeter which has connections to headphones or an external scaler. The counter only responds to thermal neutrons, so the moderators and attenuator slow down incident fast neutrons, allowing the counter to measure them once they thermalize.[6] The AN/PDR-70 is usually calibrated in counts per minute (cpm) per mrem per hour with an accuracy of +/- 20%. In this comparison, an AN/PDR-70 was used with data being fed to an external scalar-timer calibrated at 110 cpm per mrem per hour.

The polyethylene moderator in the AN/PDR-70 allows the radiac to measure fast neutrons but also causes a problem since it distorts the neutron field for any other detectors in the room. To account for this phenomenon, separate timed counts were made with the normalizing detector measuring the field intensity both with and without the AN/PDR-70 in position in order to evaluate the amount of distortion on the field and to develop a correction factor for possible use with the data acquired with the remmeter. This procedure is discussed in more detail in Chapter 9.

The AN/PDR-70 can be disassembled and its components simply used as neutron detectors. Normally it will be used fully assembled. All data referred to as AN/PDR-70 imply this configuration. By removing the outer moderator and using the remmeter, the response to fast neutrons is significantly reduced. This configuration is called the '70 guts'. By further dismantling the device by removing the attenuator and inner moderator, a person is left with a bare BF_3 probe, hence the name '70 bare'. As expected, this last configuration hardly responds to fast neutrons.

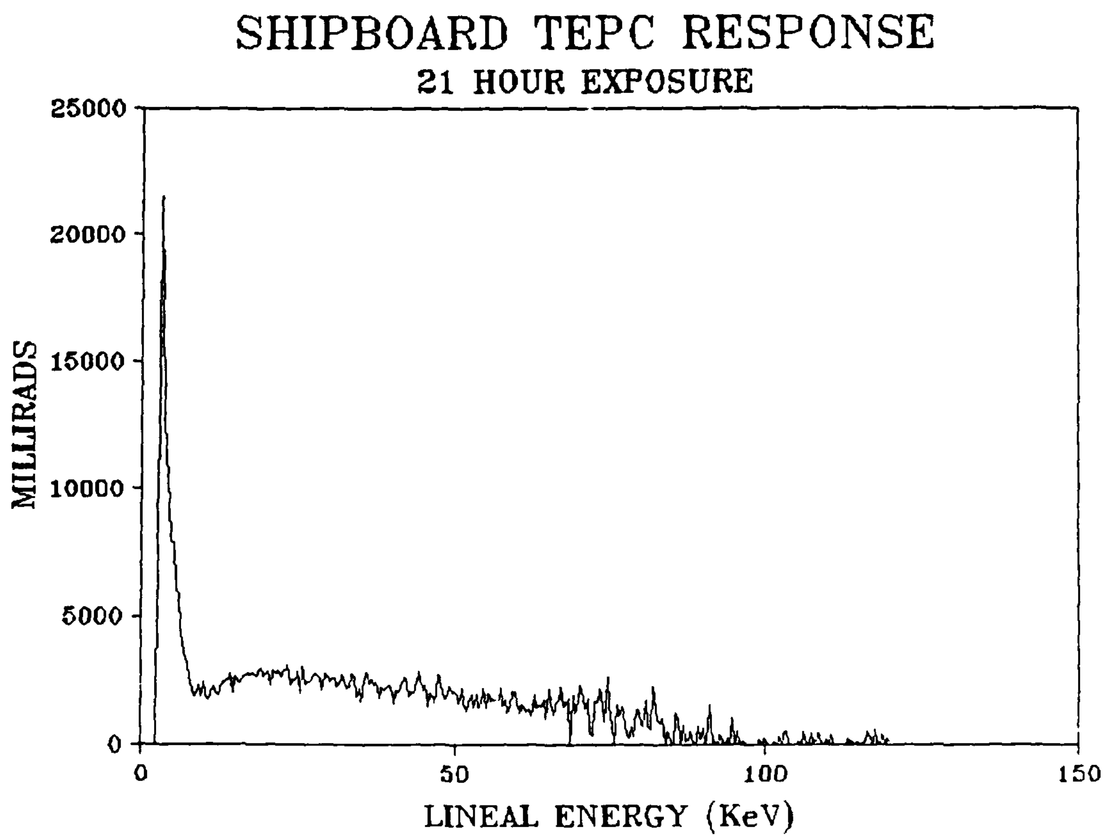
It should also be noted that for this project there were two different AN/PDR-70's in use. One was constantly in the test cell as a normalizing beam counter. The second one served as a comparison detector for the experiments. The primary reasons that the AN/PDR-70 was chosen as the normalizing detector are that it has a stable output and that it is easily read and used. Field distortion resulting from this AN/PDR-70 was minimized by separating the 70 as far as possible from the comparison detectors.

4. TISSUE EQUIVALENT PROPORTIONAL COUNTER

Technology for the Tissue Equivalent Proportional Counter (TEPC) has been around for many years, but the refinements needed to make it a useful dosimetry device have only been made in the last few years.[7,8] The device uses a sphere filled with a gas whose elemental composition is very similar to that of human muscle tissue. By adjusting the gas pressure, different masses of tissue can be simulated. Neutrons incident on the detector strike the gas and create recoil protons which cause ionizations in the gas. The electrons formed through ionizations are collected at a high voltage anode which produces an electrical signal proportional to the energy deposited by the incident neutron. The detector has a good fast neutron response, but it is also sensitive to gamma rays with the difference being that gammas produce recoil electrons instead of recoil protons. The detector output is sent through an amplifier, an analog to digital converter (ADC), and into an MCA which produces a spectrum of the number of interactions versus the lineal energy of the incident radiation. The TEPC is calibrated with an internal Curium-244 alpha source that generates a 81.72 Kev/micron photopeak when exposed. This internal source is equipped with a shutter that allows the source to be exposed for calibration in certain detector orientations and closed for experimental measurements in other orientations.

At the U.S. Naval Academy, data from the MCA are fed into a computer software package designed by Roger Hilardes '87 for his Trident Scholar Project about the TEPC.[7] This software computes absorbed doses, quality factors, dose equivalents, and dose rates for the input data. A second software package by Hilardes gives the TEPC's neutron spectrum as shown in Figure 4.1. Further details of this software are presented in Appendix C. The shipboard exposure measured in Figure 4.1 is discussed further in Appendix D.

Figure 4.1 - TEPC Measured Neutron Spectrum



To insure accurate data with the TEPC, a parametric study of the detector voltage, energy spectrum, and background corrections were made. A voltage of 500 volts was chosen to increase the system resolution of the MCA and reduce the effect of background radiation on the detector response. The neutron dose was found by integration of the spectrum only between channels 5 and 1024 to cut out the gamma data in the first several channels. The effects of correcting for background radiation were also found to be significant because regardless of detector orientation, the internal source always leaked a few alpha particles. Tables 4.1, 4.2, and 4.3 show the results of these parametric studies. All work was completed using a pressure in the detector that simulated one micron of tissue.

Table 4.1 shows the effect of applied voltage on TEPC operation. At 550 V the calibration peak at channel 464 spreads out the spectrum to the point where some higher energy data are being lost. Between 550 V and 600 V the detector became saturated and spontaneously filled all of the channels with counts. This is highly undesirable because it ruins the tissue equivalent gas. A setting of 500 V was used to maximize the energy per channel without cutting off any of the relevant high energy data.

Table 4.1 - TEPC Operating Voltage Response

TEPC Response to Cf-252 vs Normalizing AN/PDR-70 = 1

Detector Voltage	Quality Factor (Q)	Relative Response	Cm^{244} Alpha Peak Channel
400	10.4	.395	101
450	8.6	.422	169
500	7.8	.445	282
550	8.1	.517	464
600	saturation		

Proper use of the TEPC required that a correction be made for background radiation present. The primary contribution to this effect was leakage of alphas from the internal Cm^{244} source. Counts were taken both with and without the sources present and the counts received without the source present were subtracted from the ones received with the source to isolate the contribution of the source. Table 4.2 shows that this correction has a much smaller effect at a voltage of 500 V than at lower voltages.

Table 4.2 - TEPC Background Radiation Effect

Background Correction Effect on Dose and Quality Factor (Q)
Cf-252 Source, Corrected dose = 1 mrem

Detector	Cf^{252} Source Only		Source + Background	
Voltage	Q	Dose(mrem)	Q	Dose(mrem)
400	10.4	1.0	8.5	1.37
450	8.6	1.0	8.2	1.19
500	7.8	1.0	8.1	1.14
550	8.1	1.0	8.2	1.13

The TEPC is sensitive to both neutrons and gammas, so one needs to eliminate the gamma contribution to accurately determine a calculated dose with the TEPC. Previous work[7,8] discriminated neutrons from gamma dose by integrating channels with a lineal energy between 10 KeV/μ and 100 KeV/μ . This corresponds to integrating the MCA data between channels 20 and 250 to obtain the neutron spectrum. A sensitivity analysis, shown in Table 4.3, was performed on the collected data to study the effect of integration limits on the quality factor (Q) and dose rate. This analysis indicated that the channel integration limits should be carried out between 5 and 1024 to measure all of the neutron dose. These integration limits yield a quality factor of 9.1 which is in close agreement with the ICRP standards of 9.4 for this neutron source.[8]

Table 4.3 - TEPC Integration Range Effect

Effect on Dose Rate and Quality Factor

Integration Limits	Q	Dose Rate(mrem/hr)
20 - 200	8.375	.2884
20 - 250	8.914	.3284
20 - 300	9.112	.3413
20 - 350	9.264	.3507
20 - 1024	10.8212	.4335
15 - 250	8.565	.3340
10 - 250	7.900	.3444
5 - 250	7.476	.3513
5 - 10	2.057	.0095
5 - 20	2.265	.0235
5 - 1024	9.090	.4564

5. BUBBLE DOSIMETER

Bubble dosimeters are based on relatively new technology, but are simple to use and are non-albedo devices. They were evaluated in a Trident Scholar project by Eric Reilly '89[9] and their response to fast neutrons has been a key focus of this research. The bubble dosimeters used in this test consist of small vials filled with a clear polymer. Interspersed in the polymer are superheated freon droplets. As a neutron interacts with the dosimeter, the neutron creates a recoil proton which can then strike a droplet and may add sufficient energy to vaporize it and almost instantly form a visible bubble in the polymer. The bubbles formed remain fixed in place by the polymer and can be counted either manually or with an optical bubble reader. After each use the dosimeters can be reset and reused by placing them under several hundred psi of hydrostatic pressure for a short period of time (i.e. 20 minutes to an hour).

There are many factors that have to be considered to produce accurate results with these dosimeters. The detectors are initially sealed with a freon overlay that pressurizes the liquid enough to prevent vaporization and bubble formation. Once the detector seal is removed, the bubble dosimeter has a limited lifetime before it becomes unusable. Response of the devices changes with temperature, so a constant temperature environment is needed for good results. The formation of bubbles in the detector is not entirely instantaneous, so to insure an accurate bubble count, the measurements have to be taken after approximately fifteen hours to allow for complete bubble formation and growth. Even with these factors under control, there is enough deviation between different dosimeters that in order to receive accurate results, detector measurements were made with batch averages over six to twelve dosimeters.

The exact composition of the superheated liquid can alter the sensitivities of the detector. For these comparisons, two different types of detectors were evaluated. The first was the BD-100R developed at the Chalk River Laboratory in Ontario.[10] It proved to be a very reliable detector that was sensitive to both fast and epithermal neutrons. The second was the experimental BDS-10000[11] that was designed to respond only to neutrons of 10 MeV or above. The BD-100R was used throughout this research while the BDS-10000 was only evaluated for 14 MeV neutrons.

An optical bubble counter from the Chalk River laboratory was used to read the bubble dosimeters accurately and quickly .[12] It utilized a camera that took a digitized picture of the vial and then used algorithms to locate and count each bubble present. This method did present some problems. The counting algorithms required a user input of the threshold size of an object in the picture to be counted as a bubble. Also, the camera was unable to distinguish between bubbles that lined up, one behind the other, in the line of sight of the camera. To account for these problems, bubble counts were taken with the dosimeter at four different angles to the camera. An average value of the bubbles counted was then compared to the actual number counted by hand and the threshold input was adjusted accordingly to insure the computer counted all of the bubbles without also responding to system noise. Once calibrated in this fashion, the bubble reader was an invaluable aid to this research.

6. THERMOLUMINESCENT DOSIMETER

The DT-648 four chip lithium fluoride thermoluminescent dosimeter (TLD) is the current standard for radiation dosimetry in the Navy.[13] The TLD is an albedo device that responds well to thermal neutrons but can be used for personnel monitoring or for area monitoring of fast neutrons when used in conjunction with polyethylene or plexiglas blocks which serve to moderate any fast neutrons in the field. It is designed to be attached next to a persons' body to get the proper albedo response. The device consists of a card that contains four small chips that measure neutrons, gamma rays, beta particles, and X-rays. Each of these chips emits a varying amount of light, proportional to its radiation exposure, when heated. The devices used in this study were evaluated by the Naval Medical Command's Dosimetry Center at Bethesda, Maryland.[14]

The TLD was used in four different albedo modes for these comparisons. First, the Buford Area Monitor (BAM) was used as a method for neutron detection for some of the TLD's. It allowed the dosimeter cards to be completely enclosed in a lucite cylinder for accurate measurements. The BAM has a neutron response similar to the AN/PDR-70. Secondly, a Department of Energy (DOE) 40cmx40cm phantom with TLD's attached and centered was used. The DOE phantom is a large plexiglas block which simulates the dose received by a person. The third method was to place the TLD's on a 15cmx15cm polyethylene block which is referred to as a posted area monitor. The TLD's were also used in an off-body or non-albedo manner. The different configurations tested allow fast neutron correction factors to be determined for several different TLD uses.

In collecting data with the TLD's, several factors had to be accounted for to receive accurate results. Statistical uncertainties in the dosimeters were estimated by taking measurements in batches of four TLD's. The plexiglas and polyethylene used for albedo measuring provided a significant disruption to the neutron field and a correction factor for this flux disturbance was investigated. A brief experiment to determine this correction factor for use with the normalizing detector was performed and is described in Chapter 9. Some of the tests required a cadmium shield mounted to the DOE phantom to eliminate the thermal neutron contribution and make their results comparable to other dosimeters in the study. Incorporating these corrections into these tests made accurate data collection possible.

7. TRACK ETCH DOSIMETER

The Columbia Resin - 39 (CR-39) track etch dosimeter uses a set of small organic plastic insulator plates that respond to recoil neutrons by producing damage tracks on the plates. It is designed as a non-albedo device. To read the devices, an electrochemical etching process is used to enlarge the tracks for counting. This device is known to have a good response to fast neutrons but will underrespond to thermal neutrons.[15] All of the readings for these devices were done by the Reynolds Electrical & Engineering Co., Inc. in Nevada.[16]

The CR-39 measurements were done in batches of four to eight to reduce the statistical uncertainty in the dose readings. The devices were suspended at an equal distance from the sources and pointed at the source. The main problem with using the CR-39's was their directional response. It was important to insure that the correct side of the detector was facing the source to receive accurate results. Field distortion effects were not a concern for these devices.

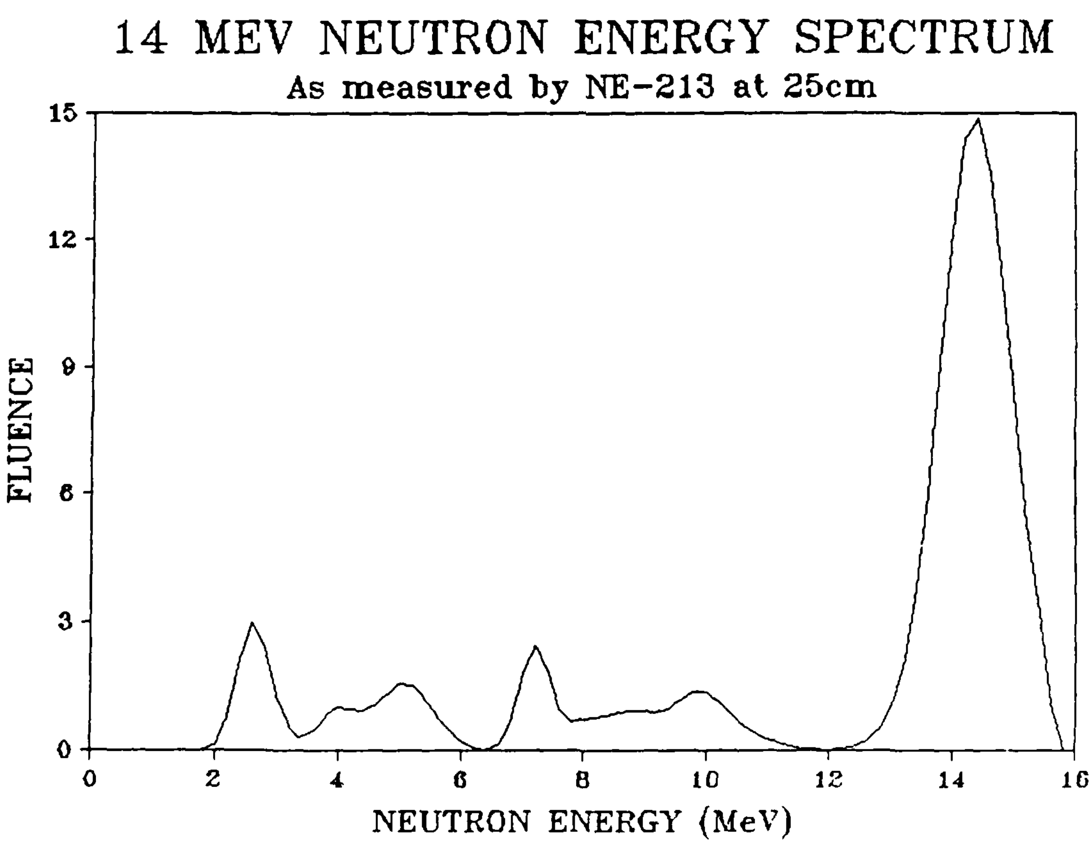
8. NE-213 SCINTILLATOR

The NE-213 liquid scintillation detector is a spectrum sensitive device that responds to both neutrons and gammas. It has a 1 MeV neutron threshold, so it only measures fast neutrons. It consists of a casing filled with an organic material that produces light, which can be related to the energy of the incident radiation. A photomultiplier tube then converts each light pulse into analog electrical pulses that can be used to determine the rise time and energy of the incident radiation. The detector signals are processed by an ND9900 dual parameter analyzer that creates a three dimensional graph of rise time, energy, and counts. High gain and low gain amplifiers are used in the data processing which renormalizes and unfolds the data to yield an absorbed dose measurement and a plot of the energy spectrum measured. This system operation has been described in detail by Fischahs.[17]

The three dimensional output produced by the NE-213 presents the possibility for very accurate radiation measurements. In using the device, it was necessary to have many independent factors under control. In taking data, the high and low gain amplifier settings used, the lower level discriminator, the power supply voltage, sodium-22 calibration changes, and energy binning were all examined to determine their effect on the detector output.

Among the most sensitive items seemed to be the power supply. It needed to be energized well ahead of detector usage to allow an hour or so for the system to 'warm up' and stabilize before data were collected. An initial calibration of the energy per detector bin done with a Sodium-22 source is required. Also important was the proper definition of regions of interest for neutron and gamma data. For the unfolding process to separate neutrons and gammas, the user is required to define distinct regions of interest (i.e. channels in the x-y plane) that represent either neutrons or gammas. This allows the computer system

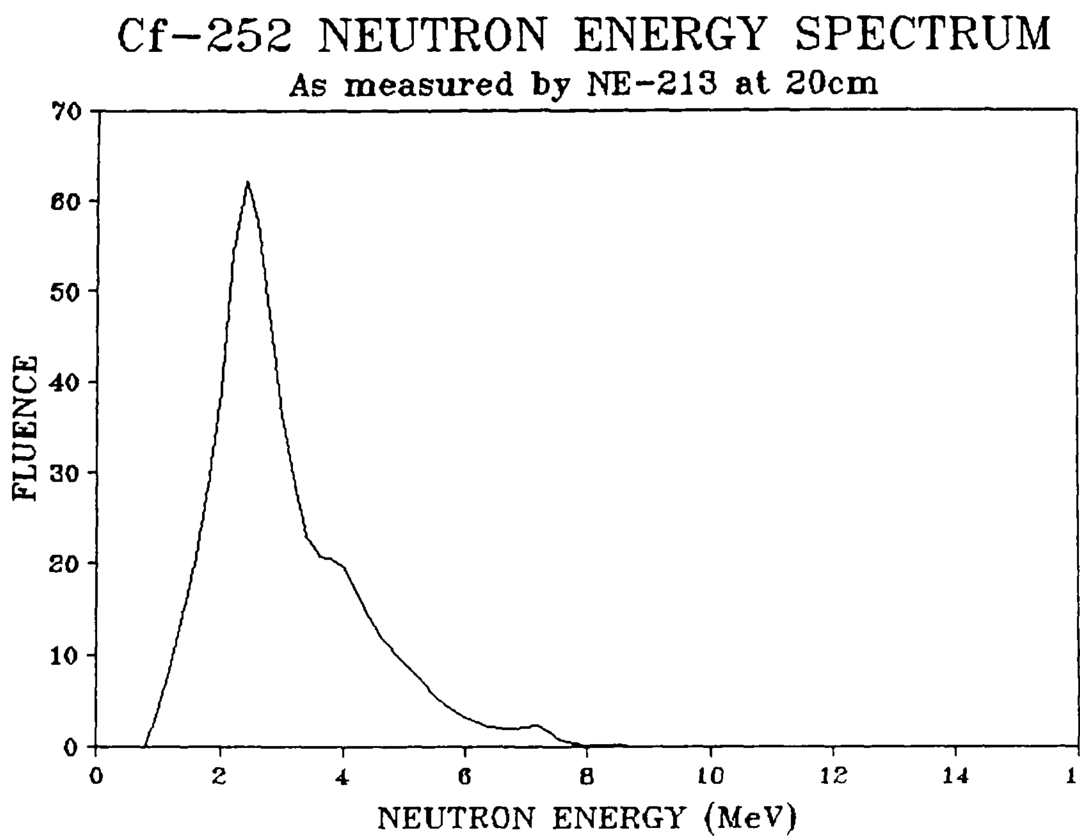
Figure 8.1 - NE-213 Prediction of 14 MeV Spectrum



to make separate data bins for neutrons and gammas. These binned data can then be corrected for lower level discriminator effects and finally run through the unfolding process to produce a dose result.[18]

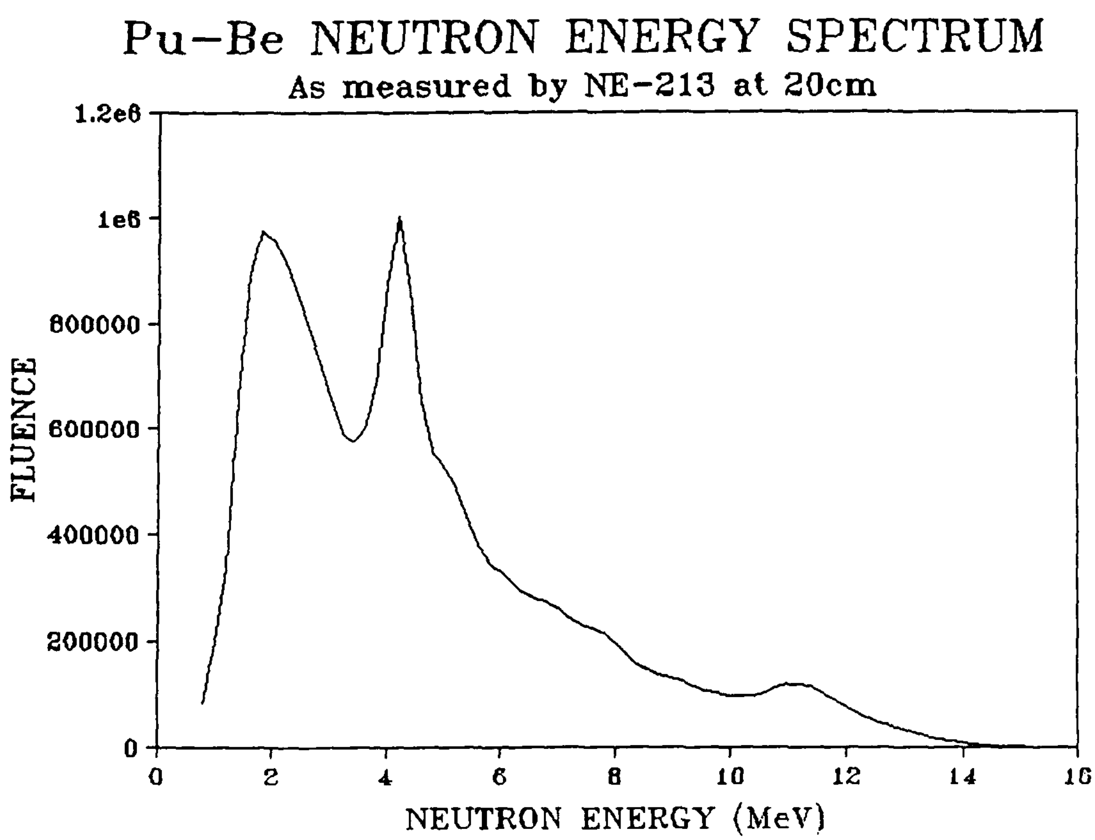
All measurements with the NE-213 were repeated several times to determine the statistical uncertainty. There was no significant flux disturbance caused by this detector, so no correction needed to be considered. The NE-213 also provided graphs of the energy spectrum measured that were used to confirm the accuracy of the unfolding code.

Figure 8.2 - NE-213 Prediction of Cf-252 Spectrum



Figures 8.1, 8.2, and 8.3 show the predicted neutron spectrum for each of the sources used in this research. Figure 8.1 shows a correct prediction of a 14 MeV peak from the USNA generator. The small peaks at lower energies are due to statistical fluctuations in the unfolding process and do not necessarily indicate a problem with the unfolding code.[19] Figure 8.2 shows the expected 2 MeV peak for a Cf-252 source. Figure 8.3 predicts the 2 MeV and 4 MeV peaks for the Pu-Be source as expected.

Figure 8.3 - NE-213 Prediction of Pu-Be Spectrum



9. RELATIVE RESPONSE

The primary focus of this research was the evaluation of the relative response of each of the aforementioned devices to fast neutron radiation and especially the 14 MeV source available at USNA. The intention was to measure each device's response to an identical neutron spectrum. By comparing the 14 MeV measured response to the measured response for each device to Cf-252 and the Pu-Be source, for which each device is well characterized, an accurate understanding of the sensitivity of each detector to 14 MeV neutrons can be achieved.

Several factors were taken into consideration to ensure the accuracy of these comparisons. To ensure that each device was exposed to the same neutron field, the measurements were done one at a time with each comparison device at the same location in the room to reduce any field distortion effects caused by the presence of other devices. This procedure also insured that the amount of room return that each device was exposed to was a constant. MCNP simulations were used to determine the detector placement to insure a large direct contribution to the dose and an adequate dose rate for counting. Each device was tested several times to estimate the statistical variance of each detector's response.

The comparison of detector measurement results requires certain special considerations. The Cf-252 and Pu-Be sources provided an essentially constant dose rate to each detector, so the dose rates measured with each device were compared directly for these sources. A constant dose rate between experiments could not be guaranteed with the 14 MeV neutron generator, so a comparison of dose received per normalizing beam count was made. This procedure introduced a possible inaccuracy in that those devices that created a distortion of the neutron flux would alter the reading of the normalizing beam counter. The AN/PDR-70, BAM, DOE phantom, and the posted block are likely to cause a significant field distortion. While no simple method was available to correct for this effect, a simple experiment was run to provide insight into the extent of this field distortion. Each detector with a significant field distortion effect was evaluated. With the neutron generator at a constant output, equal timed counts of the normalizing beam counter were made at a constant generator output both with and without the comparison device at its test location. The ratio of the counts measured indicated the amount of effect the field distortion had on the beam counter. Results of this test are presented in Table 9.1. Table 9.1 shows that the presence of all devices except the AN/PDR-70 appeared to have a small effect on the field. The AN/PDR-70's effect was found to be a significant effect with a ratio of 1.92 For future experiments, it is planned to move the beam counter to a location in the USNA test cell where field distortion effects with the AN/PDR-70 are lessened.

Table 9.1 - Field Distortion Experiment

Device	Counts With	Counts Without	Distortion Ratio
AN/PDR-70	18198	34887	1.92
BAM	47578	48391	1.02
DOE Phantom	52867	48391	0.92
Posted Block	54868	48391	0.88

Some of the devices required certain assumptions to arrive at the dose or dose rate needed for the detector comparisons. The NE-213 gives an absorbed dose measurement instead of the dose equivalent that is measured with the other devices. To convert the NE-213 dose, the following equation was used:

$$\dot{H} = \dot{D} * Q \qquad (9.1)$$

where:

$\dot{H}$ = Dose equivalent rate in rems/hour

$\dot{D}$ = Absorbed dose rate in rads/hour

Q = Quality factor

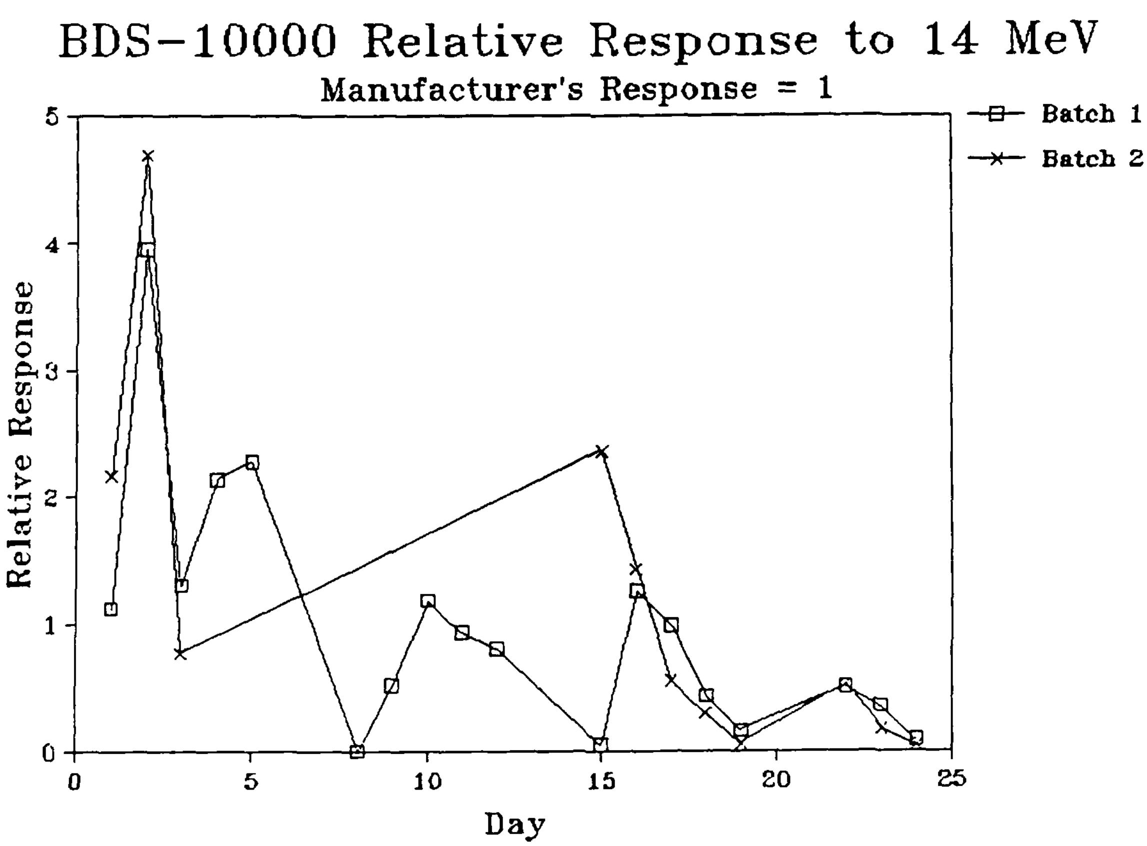
The quality factors used in equation 9.1 were generated by the TEPC data for each source. Table 9.2 lists the quality factors used in equation 9.1 by source. The MCNP code solved for neutron fluence which was then modified with

Table 9.2 - Quality Factors Used

Source	Quality Factor
14 MeV	7.90
Cf-252	7.74
Pu-Be	7.64

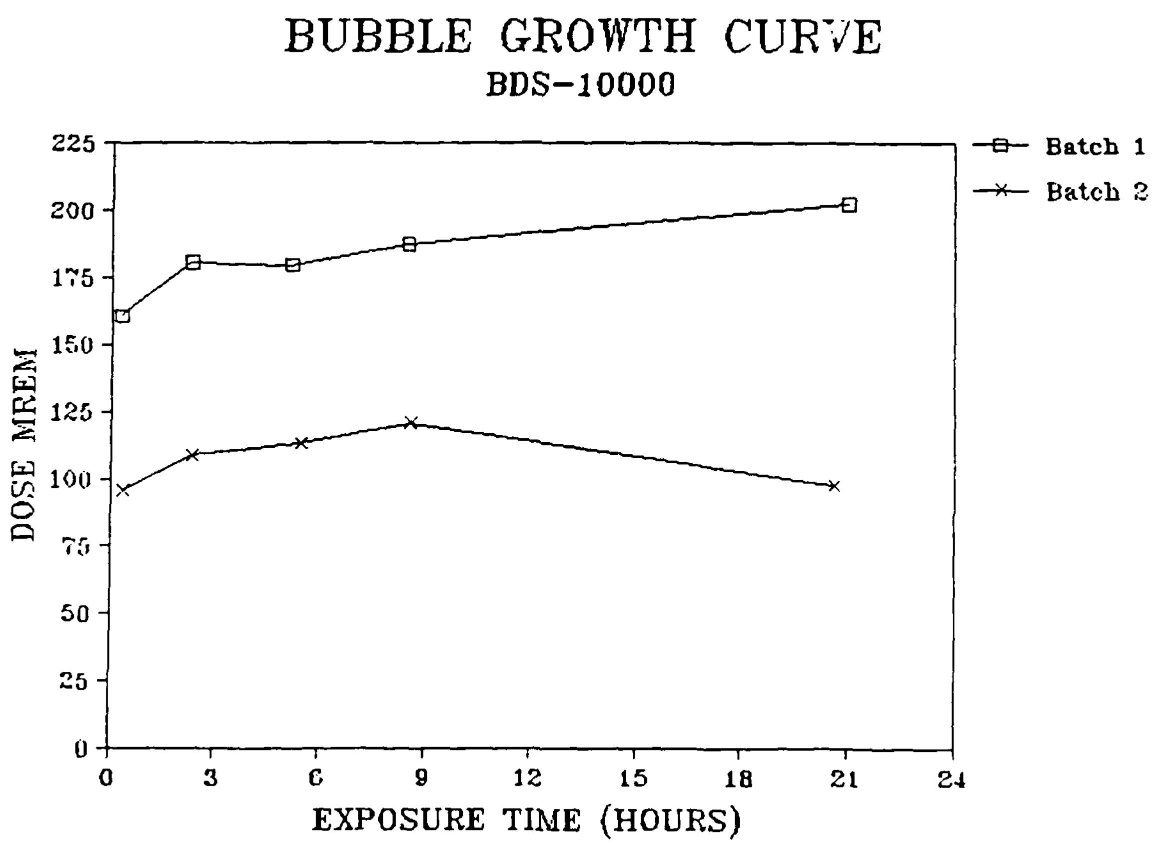
ANSI flux to dose conversion factors to compute the dose predicted. The TLD were processed using the Navy's unknown spectrum correction factor instead of specific factors for the sources or albedo conditions present.[14] In comparing the device responses, the response of the AN/PDR-70 was set equal to 1 and all other detectors were normalized to it. This is not meant to imply that the remmeter gives the true dose. This normalization was chosen because of the well characterized response of the AN/PDR-70 and to get a better perspective on the each device's response. Slightly different results would be obtained if any of these these assumptions were altered.

Figure 9.1 - BDS-10000 Response vs Time



A special case among the detectors is the BDS-10000 bubble dosimeter. It was being evaluated for its 14 MeV response and life expectancy. A batch of the dosimeters were run through a cycle of daily irradiations and repressurizations for a period of three weeks to characterize their sensitivity and useful lifetime. As can be seen in Figure 9.1, the response of these devices varied unpredictably with each reuse. This type of response would be unsuitable for dosimetric use. However, one advantage the BDS-10000's were found to have was their short response time compared to other bubble dosimeters. Figure 9.2 shows that these dosimeters reached their equilibrium dose

Figure 9.2 - Bubble Growth Curve



readings in a few hours following exposure. Other threshold bubble dosimeters were found to require 10 to 15 hours to come to equilibrium before being read. The BDS-10000 proved unreliable and was recommended for further development before any further testing of the device.[20]

The results of the detector comparison experiments are graphically displayed in Figure 9.3 and are shown in Table 9.3. The response of each device was normalized with the AN/PDR-70 defined to be unity. The field distortion effect was accounted for by multiplying the other detectors by the AN/PDR-70 ratio developed in Table 9.1. Generally

Figure 9.3 - Detector Comparison Results

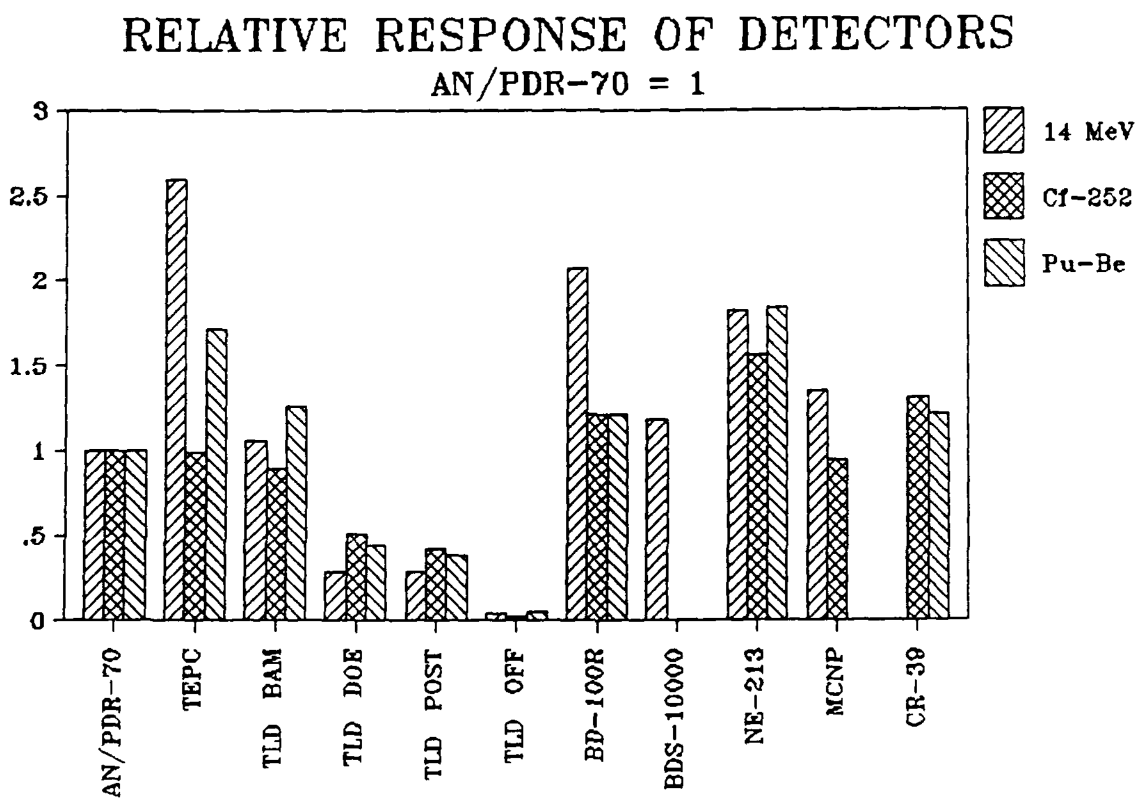


Table 9.3 - Detector Response Comparison

DETECTOR RESPONSE COMPARISON

USNA Test Cell (AN/PDR-70 = 1)

	14 MeV		Cf-252		Pu-Be	
	Ave	StnDev	Ave	StnDev	Ave	StnDev
AN/PDR-70	1.000	0.068	1.000	0.024	1.000	0.072
TEPC	2.595	0.089	0.985	0.005	1.714	0.012
TLD BAM	1.059	0.044	0.888	0.066	1.261	0.089
TLD DOE	0.284	0.037	0.502	0.016	0.442	0.028
TLD POST	0.288	0.047	0.419	0.025	0.384	0.011
TLD OFF	0.041	0.032	0.025	0.021	0.048	0.028
BD-100R	2.074	0.258	1.220	0.172	1.215	0.151
BDS-10000	6.959	7.513	------	------	------	------
NE-213	1.180	0.030	1.563	0.069	1.836	0.036
MCNP	1.817	------	0.939	------	------	------
CR-39	1.351	0.361	1.307	0.242	1.201	0.227

the trends are consistent with each detector for each of the sources. The TLD's tend to respond less because of their primary response to thermal neutrons. This was especially evident with the offbody TLD's which hardly responded. They had a very high standard deviation since the TLD only responds well to thermal neutrons. The CR-39 and TEPC tend to respond higher because they are more sensitive to the fast neutrons. The CR-39's, however, suffered from a larger standard deviation than the TEPC. It is important to note that the NE-213 gave the expected energy spectrum responses for all of the fast neutron sources. The BDS-10000 results seem to show considerable overresponse initially when compared to the manufacturers' claims. This fact combined with the very large standard deviation indicate that this device is unsuitable at the present time. The BD-100R's responded well but had a somewhat higher standard deviation than most of the devices. The MCNP code seems to be consistent with the actual detectors' responses. Most of the results for the Cf-252 and Pu-Be sources are reasonably similar because their energies are close together.

10. ROOM RETURN

An investigation into the effects of room return on dose measurements was done to distinguish between the dose caused by 14 MeV neutrons and the dose received from lower energy neutrons that were scattered off of the test cell walls. The responses of the TLD, CR-39, and the AN/PDR-70 in several modes were tested. Each device was irradiated, one at a time, with the neutron generator at distances of 25, 75, and 150 cm. The normalizing AN/PDR-70 was also present to allow a correction for the slight variations of generator output with each run. A Cadmium cover was positioned between each device and the generator to remove the thermal neutrons in the spectrum.

The three position data for each device were normalized to mrem/70 ct and then fit to an inverse parabolic curve of this form as suggested by G. Riel:[21]

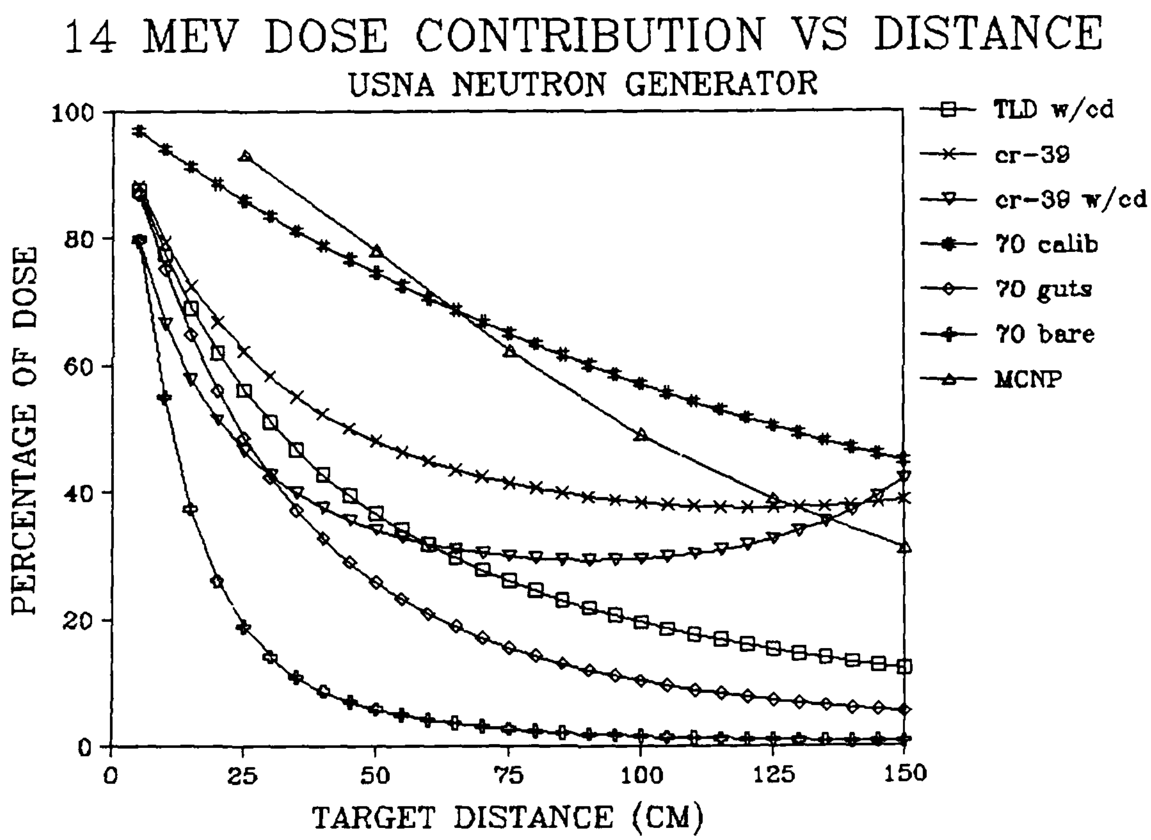
$$\text{Dose}/70\ \text{ct} = A + B/r + C/r^2 \qquad (10.1)$$

The C/r^2 indicates the direct 14 MeV dose contribution and the A + B/r term signifies the room return contribution to the dose measured. Table 10.1 lists the A, B, and C coefficients for each device. Figure 10.1 shows the 14 MeV contribution for each device and compares it to the MCNP prediction.

Table 10.1 - Room Return Coefficients

Device	$A \times 10^5$	$B \times 10^3$	C
DT-648 TLD w/Cd	10.91	22.87	0.82
CR-39	-24.90	61.30	2.27
CR-39 w/Cd	-47.90	86.93	1.63
AN/PDR-70 w/Cd	1.27	5.71	0.93
AN/PDR-70 guts w/Cd	10.47	4.68	0.17
AN/PDR-70 bare w/Cd	6.21	0.22	0.01

Figure 10.1 - 14 MeV Contribution to Device Response



The C coefficients were compared and used to determine 14 MeV correction factors (K) for each of the devices using the formula:[21]

$$K = \frac{(70\ \text{Response})}{(\text{Device}/70) * (\text{ICRP Correction})} \qquad (10.2)$$

where:

Device/70 = the ratio of C coefficients of comparison device and the comparison AN/PDR-70

70 Response = the ratio of C coefficient of the comparison AN/PDR-70 divided by the calibrated response of the AN/PDR-70

ICRP Correction = the ratio of the AN/PDR-70 response to the actual dose (.42)

The ICRP correction response was taken to be .42 as described in reference 22. Correction factors determined are listed in Table 10.2. These factors are ratios of the actual field strength to the strength measured by the device. They allow for the accurate determination of the actual dose received at a point by multiplying the dose a device measures by the appropriate correction factor.

Table 10.2 - 14 MeV Correction Factors

Device	K
DT-648 TLD w/Cd	12.67
CR-39	0.90
CR-39 w/Cd	1.25
AN/PDR-70 w/Cd	2.19
AN/PDR-70 guts w/Cd	11.77
AN/PDR-70 bare w/Cd	198.93

11. CONCLUSION

This research project has achieved several important objectives. The response of bubble dosimeters and standard shipboard neutron detectors exposed to fast neutrons and 14 MeV Neutrons in particular has been evaluated. Throughout the comparisons, the BD-100R has responded favorably when compared with other neutron detectors. Where possible, correction factors for 14 MeV radiation have been determined. These are site independent correction factors that can be applied for all devices exposed to a 14 MeV field with their accuracy being a function of the amount of room return present. An additional source of dose information is also available from the MCNP and SAND II computer codes which have been evaluated and determined to provide reasonable dose predictions for experimental purposes. Finally, measurements and calculations of the source spectra have been done which can provide valuable information for refining dose equivalence calculations.

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APPENDIX A - DEFINITIONS

The technical nature of this report requires a basic understanding of certain terms only used in nuclear science or whose meanings change when used in this field. For the reader's convenience, several of the important terms are defined here for easy reference.

Absorbed Dose - A measure of the energy deposited by radiation, within an object, units of rads

Activation Foils - Extremely pure foils of a certain element that can be easily irradiated and counted

Albedo device - A dosimeter that requires a moderator, such as human tissue, to measure dose correctly

Analog to Digital Converter (ADC) - A device which converts an analog pulse into a digital pulse

Dose Equivalent - Absorbed dose times the quality factor, a measure of the biological damage done to an object by radiation

Dosimeter - A radiation detection device that can be easily worn or carried on a person

Differential Spectrum - A plot of the neutron fluence or flux as a function of neutron energy

Fast Neutron Spectrum - An energy spectrum that is only concerned with fast neutrons (i.e. 1 KeV or greater)

Fluence - The time-integrated flux of particles per unit area, units of n/cm^2

Flux - Intensity of radiation in an area in units of $neutrons/cm^2/sec$

Lineal Energy - Energy deposited per unit path length of a particle traveling through a spherical detector

Multichannel Analyzer (MCA) - A device which bins digitized data signals based on their size. MCA's used in this report had 1024 or 4096 bins

Non-Albedo Device - A dosimeter that does not require a moderator to slow down the fast neutrons in order to measure dose correctly

Normalizing Detector - A detector used as a reference to measure other detector responses against

Point Detector - An ideal detector in which all of the response to radiation is concentrated at one point

Proportional Counter - A detector that creates electrical pulses proportional to the energy of the radiation received

Remmeter - A device that measures dose equivalent

Rise Time - The time it takes an electrical pulse to rise to 90% of its maximum pulse height.

Saturated Activity - The decay rate of a sample if irradiated for an infinite time period

Test Cell - USNA nucleonics laboratory (room R73 in Rickover Hall)

Thermal Neutron - A slow neutron with an energy less than .1 eV and is condidered to be in energy equilibrium with the surrounding atoms

Quality Factor - A factor by which absorbed dose is multiplied to obtain dose equivalent; used in radiation protection to account for the fact that equal absorbed doses of radiation can have different biological effects

APPENDIX B - GOULD COMPUTER SYSTEM USE

The U.S. Naval Academy's Gould Computer System has the MCNP and SAND II codes loaded on it. Access to the system is available by contacting computer services to get an account number and password. Once connected to computer via the fishnet server, the user of Gould can access the nuclear engineering computer codes as desired by following the procedure outlined below:

#	Prompt	Input	Comments
1	@	c gus	connects to gould system
2	Login	xxxxxxx	
3	Password	xxxxxxx	
4	%	cd ~harle/nelson	connects to MCNP directory
4	%	cd ~harle/sand	connects SAND II directory
5	%	cp oldfile newfile	copies oldfile to newfile
5	%	more filename	lists filename for viewing
5	%	vi filename	line edits filename
5	%	mc filename	runs filename on MCNP
5	%	sand filename	runs filename on SAND II
6	%	logout	exit gould system

At USNA, the MCNP code uses the input filename 'Gen' for the 14 MeV model and 'Gen1' for the Cf-252 model. The output filename is always 'Outp'. The SAND II code uses an input filename of 'Trident' and has output filenames 'Sand.out' for the numerical results and 'Plotout' for the graphical results. The 'Plotout' file requires a postscript capable printer to produce the graphs. Computer services should be contacted for information pertaining to the use of postscript printers.

Table B.1 lists the most important VI line editor commands and their functions. A more complete list is available through USNA Computer Services.

Table B.1 - Essential VI Line Editor Commands

Command	Function
h	Cursor Left
l	Cursor Right
j	Cursor Down
k	Cursor Up
x	Delete character at cursor
dd	Delete current line
rx	Replace character with an 'x'
o	Put blank below
i	Text mode
Esc	Command mode
ZZ	Exit and save work
:q!	Exit without saving work

Figure B.1 lists the input data file for the SAND II code used with the neutron generator. Figures B.2 and B.3 list the typical input data files used to generate the results presented for the neutron generator and Cf-252 sources with the MCNP code.

Figure B.1 - SAND II Input File for the Neutron Generator

```
T experimental spectrum of 14 MeV generator at 25 cm
T
T
T
T
ITERATION
10 FOILS
ZR902
ZR902  CADMIUM 2.35-3
AL27A
AL27P
AL27A  CADMIUM 2.35-3
AL27P  CADMIUM 2.35-3
MG24P
IN115G
IN115G CADMIUM 2.35-3
CU632
ACTS 5.662-18 7.172-18 1.130-18 1.684-18 1.331-18
ACTS 2.327-18 4.005-18 3.759-17 1.030-17 4.510-18
SPECTRUM TABULAR
63 POINTS
ENER 2.5-8 1-7 1-6 1-5 1-4 1-3 1-2 1-1 5-1 7.5-1 1 1.25 1.5 1.75 2 2.25 2.5 2.75
ENER 3 3.25 3.5 3.75 4 4.25 4.5 4.75 5 5.25 5.5 5.75 6 6.25 6.5 6.75 7 7.25 7.5
ENER 7.75 8 8.25 8.5 8.75 9 9.25 9.5 9.75 10 10.25 10.5 10.75 11 11.25 11.5
ENER 11.75 12 12.25 12.5 12.75 13 13.25 13.5 13.75 14
FLUX 1.318-5 2.724-5 8.850-6 4.629-6 4.624-6 4.678-6 5.162-6 5.321-6 6.643-6
FLUX 2.047-6 1.318-6 9.969-7 8.139-7 5.006-7 4.369-7 3.897-7 4.160-7 3.596-7
FLUX 2.172-7 2.790-7 1.841-7 1.299-7 1.205-7 1.204-7 9.189-8 1.309-7 1.105-7
FLUX 8.452-8 7.132-8 1.478-7 5.031-8 3.063-8 2.301-8 1.647-8 1.258-8 1.362-8
FLUX 1.417-8 2.151-8 2.719-8 9.333-9 1.359-8 1.492-8 1.871-8 1.946-8 2.238-8
FLUX 1.953-8 2.084-8 2.105-8 7.442-8 3.183-8 2.312-7 3.522-8 1.942-8 1.215-8
FLUX 1.664-8 8.197-8 2.877-9 1.439-8 1.604-10 1.575-9 0 0 1.27324-4
LIMIT 75
DEVIATION 5
DISCARD 100.0
LOW END E
HIGH END FUSION
NORM 1.0-10
PLOT, NO CARDS
SMOOTH 1
```

Figure B.2 - MCNP Input File for the Neutron Generator

```
CNP     VERSION 3A     LD= 4/28/90                          4/28/90 21:55:47
*****************************************************************************

        gen Neutron Generator
        1       0          1  2   -3   -4    13 -14 IMP:N=1
        2       1 -2.352 -3   4    5   -6    13 -14 IMP:N=1
        3       0          1  4   -5   -7    13 -12 IMP:N=1
        4       0          5  6   -3   -7    13 -12 IMP:N=1
        5       1 -2.352  1   4   -5   -7    12 -14 IMP:N=1
        6       1 -2.352  5   6   -3   -7    12 -14 IMP:N=1
        7       1 -2.352  11 7   -9   -10    13 -14 IMP:N=1
        8       1 -2.352  11 8   -9   -2     13 -14 IMP:N=1
        9       1 -2.352  3   2   -9   -7    13 -14 IMP:N=1
        10      1 -2.352  11 2   -1   -7     13 -14 IMP:N=1
        11      1 -2.352  11 14 -9   -16   8  -10 IMP:N=1
        12      1 -2.352  11 15 -9   -13   8  -10 IMP:N=1
        13      0         -17   -8 : 9 : 10 : -11 : 16 : -15 IMP:N=0
        14      0          17                      IMP:N=0

        1       PX -271.78
        2       PY -120.015
        3       PX  124.46
        4       PY  185.42
        5       PX -119.38
        6       PY  276.86
        7       PY  431.8
        8       PY -285.115
        9       PX  284.48
        10      PY  523.24
        11      PX -454.66
        12      PZ  106.68
        13      PZ -137.16
        14      PZ  228.6
        15      PZ -228.6
        16      PZ  350.52
        17      SO  900

        SDEF
        F5:N   -17.678 -17.678 0 0
        E0     2.5E-8 1.0E-7 1.0E-6 1.0E-5 1.0E-4 1.0E-3 1.0E-2 1.0E-1
               5.0E-1 0.75 1.0 1.25 1.5 2.0 2.25 2.5 2.75 3.0 3.25 3.5
               3.75 4.0 4.25 4.5 4.75 5.0 5.25 5.5 5.75 6.0 6.25 6.5
               6.75 7.0 7.25 7.5 7.75 8.0 8.25 8.5 8.75 9.0 9.25 9.5
               9.75 10.0 10.25 10.5 10.75 11.0 11.25 11.5 11.75 12.0
               12.25 12.5 12.75 13.0 13.25 13.5 13.75 14.0
        DE5    2.5E-8 1.0E-7 1.0E-6 1.0E-5 1.0E-4 1.0E-3 1.0E-2 1.0E-1
               5.0E-1 1.0 2.5 5.0 7.0 10.0 14.0
        DF5    3.67E-6 3.67E-6 4.46E-6 4.54E-6 4.18E-6 3.76E-6 3.56E-6
               2.17E-5 9.26E-5 1.32E-4 1.25E-4 1.56E-4 1.47E-4 1.47E-4
               2.08E-4
        M1      1001.04C  -.004532    $ H
                8016.04C  -.512597    $ O2
               11023.01C  -.011553    $ Na
               12000.02C  -.003866    $ Mg
               13027.04C  -.035548    $ Al
               14000.02C  -.360364    $ Si
               19000.01C  -.014219    $ K
               20000.10C  -.043546    $ Ca
               26000.11C  -.013775    $ Fe
        NPS      3000
        PRINT

```

Figure B.3 - MCNP Input File for CF-252

```
MCNP      VERSION 3A     LD= 4/29/90                              4/29/90 19:31:38
***************************************************************************************

          gen   Cf-252 model
          1       0           1  2  -3   -4    13 -14 IMP:N=1
          2       1 -2.352 -3  4   5   -6    13 -14 IMP:N=1
          3       0           1  4  -5   -7    13 -12 IMP:N=1
          4       0           5  6  -3   -7    13 -12 IMP:N=1
          5       1 -2.352  1  4  -5   -7    12 -14 IMP:N=1
          6       1 -2.352  5  6  -3   -7    12 -14 IMP:N=1
          7       1 -2.352  11 7  -9   -10   13 -14 IMP:N=1
          8       1 -2.352  11 8  -9   -2    13 -14 IMP:N=1
          9       1 -2.352  3  2  -9   -7    13 -14 IMP:N=1
          10      1 -2.352  11 2  -1   -7    13 -14 IMP:N=1
          11      1 -2.352  11 14 -9   -16  8   -10 IMP:N=1
          12      1 -2.352  11 15 -9   -13  8   -10 IMP:N=1
          13      0         -17   -8 : 9 : 10 : -11  : 16 : -15 IMP:N=0
          14      0          17                         IMP:N=0

          1       PX -271.78
          2       PY -120.015
          3       PX  124.46
          4       PY  185.42
          5       PX -119.38
          6       PY  276.86
          7       PY  431.8
          8       PY -285.115
          9       PX  284.48
          10      PY  523.24
          11      PX -454.66
          12      PZ  106.68
          13      PZ -137.16
          14      PZ  228.6
          15      PZ -228.6
          16      PZ  350.52
          17      SO  900

          SDEF  POS=-217.78 42.985 -17.16  ERG=D1
          SP1   -3 1.025 2.926
          F5:N  -197.78 42.985 -17.16 0
          E0    2.5E-8 1.0E-7 1.0E-6 1.0E-5 1.0E-4 1.0E-3 1.0E-2 1.0E-1
                5.0E-1 0.75 1.0 1.25 1.5 2.0 2.25 2.5 2.75 3.0 3.25 3.5
                3.75 4.0 4.25 4.5 4.75 5.0 5.25 5.5 5.75 6.0 6.25 6.5
                6.75 7.0 7.25 7.5 7.75 8.0 8.25 8.5 8.75 9.0 9.25 9.5
                9.75 10.0 10.25 10.5 10.75 11.0 11.25 11.5 11.75 12.0
                12.25 12.5 12.75 13.0 13.25 13.5 13.75 14.0
          DE5   2.5E-8 1.0E-7 1.0E-6 1.0E-5 1.0E-4 1.0E-3 1.0E-2 1.0E-1
                5.0E-1 1.0 2.5 5.0 7.0 10.0 14.0
          DF5   3.67E-6 3.67E-6 4.46E-6 4.54E-6 4.18E-6 3.76E-6 3.56E-6
                2.17E-5 9.26E-5 1.32E-4 1.25E-4 1.56E-4 1.47E-4 1.47E-4
                2.03E-4
          M1      1001.04C   -.004532     $ H
                  8016.04C   -.512597     $ O2
                 11023.01C   -.011553     $ Na
                 12000.02C   -.003866     $ Mg
                 13027.04C   -.035548     $ Al
                 14000.02C   -.360364     $ Si
                 19000.01C   -.014219     $ K
                 20000.10C   -.043545     $ Ca
                 26000.11C   -.013775     $ Fe
          NPS     3000
          PRINT
```

APPENDIX C - MISCELLANEOUS PC SOFTWARE USE

Several PC based computer programs and software packages were used to aid in the research. A brief discussion of these programs is presented to aid future researchers.

The TEPC.cal[7] program developed by Roger Hilardes for his Trident scholar project was used with the Supercalc 4 spreadsheet for TEPC dose calculations. It was used in conjunction with the BACKGROUND.cal file which removes the background contribution to the TEPC detector from the data file. The TEPC.cal program took the raw channel number vs number of counts data and used it along with inputs of the detector diameter (inches), energy calibration (KeV/channel), and tissue equivalent gas pressure (mm Hg) to calculate the neutron quality factor, absorbed dose (rads), dose equivalent (rems), and the dose rate (rems/hr) measured. A second program by Hilardes called Chipgraph.cal[7] took the MCA data and made a plot of lineal energy vs absorbed dose as shown in Figure 4.1. For the field test data, a portable Davidson MCA that stored data on a magnetic tape was used with an advanced basic (Basica) program called Transfer.bas which moved the collected data to the PC for analysis with either TEPC.cal or CHIPGRAPH.cal. These programs were invaluable for the TEPC research.

The majority of the data storage for the bubble dosimeter experiments was done with the Oracle relational data base management system. A specific format for data entry was developed at USNA and provided a quick means of data storage and retrieval. By creating a library of dosimeter information initially, all of the information for each dosimeter used was automatically brought up and used by simply referencing the dosimeter's serial number. This Oracle format simplified the data evaluation process and provided an organized storage capability. The bubble dosimeter data on Oracle has been forwarded to Oak Ridge National Laboratory and entered into a permanent data base with bubble dosimeter data from other researchers.

APPENDIX D - FIELD TEST

During this research, the opportunity presented itself to take detector comparison data in a typical shipboard radiation environment onboard a Trident submarine. Neutron radiation measurements were made as part of the Navy's Superheated Liquid Drop Research Team. Measurements were made using the AN/PDR-70, TEPC, TLD, and several types of bubble dosimeters. Measurements were made over a period of one week with readings being made approximately twice daily. The use of the TEPC was the first time it had been used in a shipboard environment and no problems were encountered. BD-100R and BDS-100 bubble dosimeters from the Chalk River Laboratory were compared with Apfel[23] designed devices, one (APFEL PEN) that measured dose received by the volume displacement of the bubbles in the matrix and another (APFEL A/SM) that measured dose by counting the sonic pulses occurring at bubble formation. A comparison of the dose each device measured in equivalent radiation fields is shown in table D.1.[24]

Table D.1
Field Test of Device Response Comparison

Device	Dose Rate (mrem/hr)
AN/PDR-70	.487
TEPC	.460
TLD BAM	.536
TLD DOE	.181
BD-100R	.643
BDS-100	.580
APFEL PEN	.408
APFEL A/SM	.729

These results indicate that the bubble dosimeter seems to give consistent results when compared with other neutron dosimetry systems used in a shipboard environment.